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SOME WAYS OF WATER-WATER POWER REACTOR DEVELOPMENT

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I. Peculiarities of reactor and plant design of the Novo-Voronezh atomic power station second unit

Water-water power reactors with application of water as a moderator and a coolant are represented in the USSR by two units of the Novo-Voronezh atomic power station. Its first unit of 210 Mw power is being now in the starting stage and the second one of 365MwE power is being under construction.

The first unit was described earlier [1,2,3] . Construction of this unit pursued the aim to gain an experience in design, erection and operation of the stations utilizing heterogeneous pressurized water reactors. It is natural that parameters and design solutions for this first large atomic power station were selected in some cases with a certain extent of precaution.

Design of the second stage of the Novo-Voronezh atomic power station pursues the aim, on one hand, to make a new step towards construction of atomic power plants with economic indices getting close to those of thermal power plants and, on the other hand, to make maximum use of experience gained during the first stage erection in designing and manufacturing basic equipment avoiding such new solutions realization of which would require a considerable consumption of time and means for research and design work. That is why the design of the second unit of the station should be considered as a modernized variation of the first stage within aforesaid limits.

The reactor vessel, like that of the first unit, will be close to an expedient technical maximum for transportable factory-made vessels. It will not be covered by an anti-corrosion cladding that will make it possible to enhance thickness of the basic metal as well as the rated pressure (up to 120 absolute atmospheres).

To ensure a unification, the fuel core as that of the first unit will comprise 349 hexahedral assemblies 2.5 m long and sized for a 144-mm wrench.

Cylinder -shaped fuel elements will contain sintered uranium dioxide in jackets ~.5mm thick made of either zirconium-niobium alloy or stainless steel.

To cut short the time of erection, it has been decided to use main circulating canned-type pumps which have been mastered by the industry and used in the first unit (capacity- 5250 cub m p.h., pressure head - 60 m of water column). On the same reason the fuel recharging remains to be executed with a removal of the cover; simultaneously the work is carried out to design a device for fuel recharging without removal of the cover in order to utilize such a system in the future.

Due to the fact that the removal and mounting of the cover, fuel recharging itself and the restoration of the operating state of the reactor plant may take a lot of time, these operations should be conducted as seldom as possible, i.e. a duration of the operation between two routine rechargings of fuel should be long enough. Provided an increased specific power of the reactor, it would mean a corresponding increase of the fuel burn-up and, at the same time, an increase of the initial reactivity margin which must be compensated in the operating devices of the reactor control and safety system.

A principle of the excess reactivity compensation used here is based on substituting the fuel for absorbent with an application of a fast neutron water trap. Design studying has discovered a possibility for increasing the number of these devices up to 73 assemblies distributed evenly within the fuel core in the triangular lattice points with a pitch of 294 mm.

The water-to-uranium ratio (a ratio of water and fuel material volumes) remains the same as that of the first unit (I.7). It ensures a minor negative coefficient of the reactivity, a sufficient compensation capability of the control system and a good neutron balance.

The fuel is charged in assemblies of nearly the same design and the same cross-section size as those of the first unit.

The diameter of the UO_2-pellets is 7.7 mm and the external diameter of the fuel element is 8.8 mm, each assembly can contain I27 fuel elements within a triangular lattice with a pitch of I2.3 mm. And this made it possible to increase the thermal capacity of the reactor up to ~ I400 Mw, i.e to exceed ~ I.8 times the capacity of the first unit. Further essential increase of the capacity might become useful only upon an increase of water flow rate and an alteration of reactivity compensation and recharging systems that would contradict initial design provisions.

Within a core life of ~ I.5÷2 years, three partial rechargings provided, the uranium burn-up reaches ~ 22,000 ÷ 30,000 MWD/T. Average enrichment needed for the first charge is 2.5 - 3.5 per cent and a fuel make-up of the reactor is executed by a fuel of the 3-4 per cent enrichment.

60 from 73 assemblies of the control devices are designated for compensating slow changes of the reactivity and I3 others - for safety system of the reactor.

Design of control and safety system assemblies is similar in principle to those of the first unit; nevertheless, provisions are made for a hydropiston relief of the assemblies against upward strain caused by an upstream of the coolant which hampers a reliable downward travel of the assembly toward the side of a reduced reactivity in this reactor which is characterized by increased heat rate.

Both safety and control assemblies consist of two parts, one of which is an absorber and another part contains fuel.

Usage of the fuel in assemblies of the safety system provides an improvement of the neutron balance in the fuel core and an increase of the fuel burn-up.

A decision has been made to utilize a saturated steam power cycle under a pressure of 30 abs atm, though studying has proved in principle an economic advantage of utilization of fire overheating which, however, complicates the plant design. The second unit will be provided with five turbines similar to those of the first unit; some modernization will ensure an increase of each turbine capacity up to 73 Mw. So, the electric capacity of the unit will be of 365 Mw with an efficiency of about 26 per cent.

To extract the power from the fuel core avoiding a reduction of parameters of the steam delivered to the turbines and reactor inlet water temperatures a decision has been taken for an increase of the water maximum enthalpy in the outlet of the hottest fuel assemblies up to a saturation enthalpy and even for the admission of the steam outlet quality (up to 3-4 per cent) in some streams of hot channels. An average heating of ~ 28°C and water temperature of 250°C in the reactor inlet provided, a consumption of the coolant passing through the fuel core reaches 42,000 cub m p.h. It requires an increased number of the plant circulation loops (8 comparing with 6 in the first unit). Owing to a refusal of the possibility to repair separate units during the operation of the reactor,

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the first circuit is simplified considerably and this made it possible to accomodate 8 loops in two hermetic boxes.

To maintain a needed quality of water in the first circuit, the purification system of the second unit is provided with ion-exchange filters unlike the evaporators erected in the first unit of the station; it ensures a simplified stabilization of the gas balance. The filtration is performed at the operational pressure of water.

No gas pressurizers as those of the first unit but steam ones provide a maintenance of the pressure in the first circuit that also facilitates a stabilization of the first circuit hydro-chemical conditions.

As a result of an increase in the specific and general reactor power it has become possible to gain a considerable drop in the fuel and capital components of the electric power cost.

The reactor (Fig.I) is a vertical cylinder-shaped body of about I9 m in height. It consists of a vessel (I0) and an upper removable block (I2) with a flat cover. A special cylinder-shaped pit inside the vessel contains a detachable basket (7) of 3,000 mm in diameter and 4,000 mm in height designed for an accomodation of the hexahedral assemblies with fuel elements (5).

Movable assemblies (2), designed for a compensation of the reactivity and for a reactor emergency protection, are provided with drives mounted in the upper removable block jackets.

A cover of 3350 mm in diameter and 520 mm thick is mounted on the vessel flange. Branch pipes for control system devices and for jac... of the assembly water temperature control thermocouples are welded in 85 holes of I00 mm in diameter. The coupling of the cover and the vessel is ensured by a self-sealing lock with a wedge-shaped nickel packing. As a reserve version, provisions are made for a sealing by a flexible element of the tore compensator; one end of this element is welded to the cover and another rests on the vessel flange. The tighteness here is obtained either by a bar packing of 5 mm in diameter or by welding the tore compensator to the vessel flange. The cover is secured to the vessel by 60 flange bolts through a locking ring of 4,000 mm in diameter and 800 mm in height.

Principal scheme of the power station and cooling system of the reactor is shown in Fig.2. Each circulation loop includes a steam generator, a centrifugal canned type pump, two valves with an electrical drive, a return valve and connecting pipelines with an internal diameter of 500 mm.

The steam generators designed will be of horizontal type. This will make it possible to set a heat-exchange surface lower than the vessel flange of the reactor in order to keep the circulation loops under water while recharging the reactor. That will ensure, in turn, a removal of residual heat during the reactor recharging without a special circuit by a sufficient natural circulation.

In the event of a power supply loss in the station auxilliery network the power station will be provided with a circuit of the reliable power supply comprising storage batteries and Diesel engine-driven generators with autostarts. Power supply of the circuit circulation pumps since the first seconds of a failure will be provided by the turbine generators at the expense of the turbine final run due to inertia and heat content of the system. Upon a decrease of generator revolutions lower than a normal rate the pumps are to be switched to the storage batteries with a further stop and transference to shut-down cooling by the natural circulation in the reactor circuit.

Arrangement of the turbine and deaerator section equipment is similar to that of the organic fuel-operated power stations.

2. Ways for improvement of water-water power reactors.

Improvements of water-water power reactors, to be realised by the design of the second unit of the Novo-Voronezh atomic power station, have been strictly limited by the aforesaid modernization of the first unit and that is why it is far from exhaustion of all ways for the development of water-water power reactors.

From the standpoint of a probable reduction of the power generation cost it is necessary to consider some relatively radical ways for improving the design of vessel-type water-water power reactors.

Steam generation circuit

It seems to be advantageous to keep a simpliest principle of the steam generation in evaporating "boilers" with a delivery of the saturated steam to turbines.

A generation of the energy-producing steam at the expense of water self-evaporation in the reactor circuit during water orificing can compete with the utilization of surface-type steam generators only in the cases of a low (quite inoptimal) pressure and an application of sufficiently cheap pumps with a cost (per kw of pump capacity), as calculations have proved, more than two times lower than the cost of one sq m of the heating surface.

Fig. 3 shows an increase of a relative expenditure of the energy Δn for a circulation and an increase of the cycle efficiency $\Delta\eta$ depending on the initial and final pressures in case of a replacement of a steam generator with the maximum temperature difference ϑ by an orificing plant. As it is seen, both increases are equal ($\Delta n = \Delta\eta$ i.e. the net plant efficiency is not reduced) either in case of low pressures of the steam (p_2 = 10÷15 abs atm) which are disadvantageous for large-scale energetics or in case of disadvantageously high temperature differences $\vartheta > 25^\circ C$.

The steam superheat $\Delta t_{sup.}$ in a steam generator results in an increase of cycle efficiency $\Delta\eta$ which is approximately proportionate to a product of Δt_{sup} and a heat fraction of the superheat $q_{sup} = \bar{C}_p \Delta t_{sup} / \Delta i$. As a possible superheat is proportionate to the average coolant temperature rise in the reactor Δt, an increase of the efficiency caused by an introduction of the superheat is approximately proportionate to Δt^2. As minor water heatings are optimal for water-water reactors, an effect of the overheating introduction does not, as a rule, prove its value due to a complication of the design and a rise of the steam generator cost.

An application of a two-pressure cycle enhances the efficiency proportionately to the first power of Δt that makes an increase of the efficiency more essential even with a low Δt. E.g., providing $\Delta t = 40^\circ C$, a relative increase of the efficiency will be approximately as follows:

$$\frac{\Delta\eta}{\eta} \cong \frac{1}{4} \cdot \frac{T_x \Delta t}{(T_1 + \Delta t/2)(T_1 - T_x)} = \frac{1}{4} \frac{300 \quad 40}{(500+40/2)\,(500-300)} \cong 03$$

where $T_x = 300^\circ K$ and $T_1 = 500^\circ K$ are the temperatures of cool and hot sources in an equivalent one pressure cycle.

A uniform water heating in assemblies of the reactor provided, the optimal heat Δt will increase and, as a result of this, the effect may increase ~ 1.5 times.

In case of a successful overcoming the turbine moisture separation difficulties it may happen to be advantageous, a high capacity of each cooling loop in the reactor provided, to use two incorporated in series into the first circuit steam generators of similar type which produce a saturated steam of two temperatures differred by 20–25°C.

But it should be noted that this would, to a certain extent, complicate an arrangement of the first circuit which must be simpliest and symmetric so that to minimize the volume and length of pipes and premises of the first circuit as well as to create proper conditions for the natural water circulation in the first circuit and to compensate thermal expansion of pipes.

A thermal expansion compensation is preferably to provide by free shifting the equipment, if the pipelines are rigid enough and, if possible, straight. In this case it is expedient to place the pipelines of the first circuit, mountings of the reactor, pumps, steam generator and fittings approximately at one and the same level; it is useful to secure main pipes to the top part of the reactor upper than the level of the fuel core that would simplify the erection structures and increase the reliability of the reactor vessel.

Water circulation direction in the reactor

When water in the reactor moves downward the space above the fuel core is free for the conduct of the recharging operations and an emergency insertion of the control system absorbers into the core under a combined effect of the flow and gravitation becomes more reliable.

However, there is an apprehension that a reversal and depression of water circulation may occur in the event of an emergency while switching off the power supply of the main pumps, and that it may cause an excessive overheating and damage of the fuel elements. Such conditions will unlikely cause serious consequences. But it is very difficult to prove this with a sufficient degree of trustworthiness. Now our consideration will be restricted by studying an upward motion only during which aforesaid apprehensions are practically disappeared, yet this motion causes serious difficulties for ensuring a space above the fuel core and for a reliable downward motion of the control and safety system operating devices .

A two-pass fuel core in some respects takes an intermediate position between single-pass cores with upward and downward motions of water.

In such fuel cores with water inlet and outlet in the bottom it is convenient to fix the assemblies in the central upward pass by hydropistons and a water temperature control for each assembly should be organized in the bottom only, i.e. in the outlet of the perifery (downward, hot) pass. In this case a space above the fuel core will be unobstructed for recharging without removal of the cover. A downward motion of water can be arranged in all the channels of the control and safety system.

But the two-pass fuel core contains a potential possibility for an occurrence of the power neutron instability and still excites an apprehension in respect of a reliable provision of an emergency shut-down cooling in the downward pass in the event of a sudden cessation of the forced circulation.

Yet, thermotechnical advantages, connected with an increase of critical heat fluxes (due to an increase of the rate and subcooling of water in a dangerous point) and with a possibility for an increase of the temperature rise and a reduction of the water flow rate (due to flattening the temperature in the chamber between passes), can be essential only in the case of rather high radial irregularity of the heat release which itself is unprofitable.

That is why the two-pass fuel core usually has no sufficient thermotechnical advantages which may justify an increase of the hydroresistance and a complicacy of the design.

Methods of recharging

Under conditions of a high pressure of the coolant and a channelless structure of the fuel core a fuel recharging with a removal of the reactor cover to-day is most practicable. Auxiliary operations in this case require a lot of time (E.g. see [5]) and are accompanied by an essential change in "thermomechanical" conditions of the first circuit equipment. That is why it would be advantageous from the economical point of view to perform such rechargings rarely (not more than a few times within one core life) in spite of the fact that a more frequent recharging would enhance the burnup of the extracted fuel by making use of the additionally charged fuel reactivity [4] , and simplify the problem of an excess reactivity compensation.

A reduction of the recharging duration reduces the power underproduction and provides a drop of the cost of a kwh.

Making use of an approximate relation between a possible burn-up $\rho_w(n)$ and the number of rechargings n during a core life τ_K (see [4]) $\rho_w(n)/\rho_w(n=\infty) = (1+\theta_2/n)^{-1}$ it is possible to obtain a dependence of an optimal number of rechargings n_{opt} on the auxiliary time τ_B of a water-water power reactor stoppage caused by the recharging, as well as on the pure time needed for the recharging of the assemblies $\tau_n = m\tau_1$

$$n_{opt} = \frac{1 - \tau_n/\tau_K}{\frac{\tau_B}{\tau_K} + \sqrt{\frac{\tau_B}{\tau_K}\frac{1}{\theta_2}\frac{C_K^0}{C_T^0}}} \cong \sqrt{\theta_2 \frac{\tau_K}{\tau_B}\frac{C_T^0}{C_K^0}} \qquad (I)$$

where m is a total nuber of the assemblies, τ_1 is the time needed for the recharging of one assembly;

$$\theta_2 = \frac{\int_0^R r\,dr \cdot \int_0^R \phi(r) r\,dr}{\int_0^R \phi(r) r\,dr \cdot \int_0^R \phi^2(r) r\,dr}$$

is a coefficient of the burn-up fraction reduction due to a radial irregularity of the neutron field $\phi(r)$; when $\phi(r) = J_0(\mu \frac{r}{R})$ $\theta_2 = 1.69$; C_K^0, C_T^0 and $C_э^0 = (C_K^0 + C_T^0)$ - capital and fuel components and the cost of a kwh under ideal conditions of a continuous recharging under a full capacity.

When $n = n_{opt}$, the cost of a kwh $C_э(n) = C_{min}$ is higher than $C_э^0 = C_T^0 + C_K^0$ by the following fraction:

$$\frac{\Delta C(n_{opt})}{C_э^0} = \frac{C_{min} - C_э^0}{C_э^0} = \frac{\frac{\tau_n}{\tau_K} + \sqrt{\frac{\tau_B}{\tau_K}\theta_2\frac{C_T^0}{C_K^0}}\left(2 + \sqrt{\frac{\tau_B}{\tau_K}\theta_2\frac{C_T^0}{C_K^0}}\right)}{\left(1 + C_T^0/C_K^0\right)\left(1 - \tau_n/\tau_K\right)} \cong \frac{2\sqrt{\theta_2 \tau_B/\tau_K}}{\sqrt{C_T^0/C_K^0} + \sqrt{C_K^0/C_T^0}} \qquad (2)$$

These dependences are presented in Fig.IO, and Fig.II deals with the dependence $\Delta C(n)/C_э^0$ of excess of the kwh cost over an "ideal" one on the number of rechargings n . It is clear from this dependence that a reduction of n down to $.5\,n_{opt}$ has no much effect on the increase of the energy cost.

It can be seen that a major role is played by the auxiliary time a reduction of which down to $(10^{-3} \div 10^{-4})\tau_K$ may considerably cut down the energy cost. Side by side with improvement of the recharging with the cover removal the realization of such short periods of time for auxiliary operations would stimulate a design of rechargings without the cover removal and without shut-down cooling of the reactor.

A fuel recharging without the cover removal but after a cessation of the reaction, shut-down cooling and pressure drop will be perspective unlikely as in comparison with a recharging conducted with an open cover this method will exclude only operations on repacking the cover, yet operations on actual recharging would be more complicated and, apparently, would take longer time. The reactor shut-down cooling before the recharging and a heating after it should be conducted rather slowly to avoid probable thermomechanical damages.

An "in-run" recharging requires some complicated measures, which can ensure a reduction of the water consumption through the cell of the extracted assembly, and causes a pick of the neutron field in assemblies adjacent to that extracted. Such a method of the recharging will apparently cause a loss of the reactor capacity. Conduct of recharging operations in a critical state of the reactor also causes some difficulties and considerably slows down the recharging process. An application of solid absorbers inserted from above is impossible at all; a usage of the lower drive is restricted due to inconveniences caused by the location of absorbers above the fuel core. With this aim the selection of the reactivity compensation system apparently comes to an application of liquid compensators, i.e. by changing the concentration of an absorber in the solution.

To this effect the most suitable thing would be apparently a subdivided fuel core with cells (resembling "honeycombs") formed by special partition walls which, however, take up an appreciable part of the useful volume and look like a tracery structure (Fig.6).

When a removal of the cover is not provided it is most useful to recharge fuel without a reduction of the pressure and temperature but after a shut-down of the reaction. Such rechargings can be performed in two stages: a) an extraction of assemblies from the active section to the cells of an intermediate storage (or: in the opposite way), b) a removal of assemblies from these cells through a vessel lock (or: insertion of new assemblies into the cells), and, as a result of this order, the most complicated second stage can be conducted without time limitation under the conditions of the reactor operation.

A design lay-out of this recharging is similar to that shown in Fig.6. There are movable or fixed cells inside the reactor vessel for an intermediate storage of assemblies outside the active section. These cells and the lock on the cover are distributed one above another that would facilitate kinematics of the external recharging device. Its container may be provided with a "magazine" to perform locking only once during one recharging.

When there are n rechargings during core life the number of cells of an intermediate storage should approximately make up a $1/n$ fraction of the number of assemblies in the fuel core (if the number of assemblies is 300 and the number of rechargings varies from 20 to I00, the number of intermediate cells needed should be from 4 to I6).

When a replacement of one assembly lasts τ_1 = 15-30 min, the number of assemblies is 300 and the core life duration is τ_K = 500 days, an increase of the kwh cost due to deviation from conditions of a continuous recharging with the full capacity will make up rather low value:

$$\frac{\Delta C(n_{opt})}{C_э^\circ} \cong \frac{1}{2}\frac{m\tau_1}{\tau_K} = \frac{1}{2}\frac{300(0.25\div 0.5)}{500\cdot 24} \cong (0.3\div 0.6)\%$$

Evaluating the duration of auxiliary operations (a shutdown of the reaction, a removal of control drives for ensuring a space above the fuel core and a restoration of the operating conditions) as τ_B = 5hrs, with θ_2 = 1.69 and C_K°/C_T° = I, the following result can be obtained:

$$n_{opt} \cong \sqrt{\theta_2 \frac{\tau_K}{\tau_B}\frac{C_T^\circ}{C_K^\circ}} = \sqrt{1.69\,\frac{500\cdot 24}{5}\cdot 1} \cong 65$$

and the cost of a kwh will additionally increase by:

$$\frac{\Delta C(n_{opt})}{C_э^\circ} \cong \frac{2\sqrt{\theta_2 \tau_B/\tau_K}}{\sqrt{C_T^\circ/C_K^\circ}+\sqrt{C_K^\circ/C_T^\circ}} = \sqrt{1.69\,\frac{5}{500\cdot 24}} = 2.6\%$$

When the drives are mounted beneath the fuel core the auxiliary time τ_B can be reduced that will result in the reduction of this percentage.

Assemblies holding

In a reactor operating in forced conditions the lifting force of the upward water flow can be of several times stronger than the weight of the assemblies. That is way the assemblies must be held to prevent their upward travel or releaved from the pressure drop.

If a positioning lattice is provided, the assemblies can be held by locking springs fixed in the walls of the lattice, as well as by controlled locks or spring collets attached to the tail of the assemblies to provide fixation in the bottom support grid (see Fig. 6, pos 8). These methods require either an extended space between assemblies, or an increase of the recharging device effort, or an application of appropriate devices, passing through the core, to control the locks.

A more rational method of the removal of assemblies seems to be an application of pistons connected with the tailing part of the assemblies and subjected to a reverse pressure drop as there is a chamber beneath the pistons for a leakage collection connected by a by-pass pipe with the space above the fuel core.

All the methods described give a possibility to relieve the top part of the pressure vessel from devices obstructing the fuel recharge.

Devices and drives of control and safety system

Solid absorbers traveling along the axis of the core are most frequently used in water-water power reactors. The reactors of both stages of the Novo-Voronezh atomic power station equipped by control assemblies with absorbers provided with neutron water "traps".

A replacement of compensating absorbers and safety absorbers by the fuel upon their lifting ensures a good neutron balance and increases a compensating capability of the control devices but, on the other hand, causes some difficulties when the absorbers are being promptly inserted against the direction of coolant flow. Therefore in this case it is necessary to ensure a relief of the control assemblies from the effect of the upward flow under the conditions of high heat removals rate.

Control devices of these like cause a considerable irregularity to the neutron field distribution in intermediate positions and that is why, speaking generally, it is advisable to use them for an emergency protection and a compensation of the reactivity changes caused by the temperature effect, Doppler-effect and the poisoning.

It seems to be useful to attach a compensation of the fuel burn-up to absorbers with a relatively low effectiveness as their travel would not cause marked neutron flow irregularities; it is possible, for example, to utilize flat or cylinder-shaped absorbers provided with scattering tails. When rechargings are conducted rather frequently (rechargings without a cover removal are meant), an excess of the reactivity compensating the burn-up may be low and due to this fact the total effectiveness of absorbers cannot be high as well.

A design of combined absorbers which would combine principles of absorption in "traps" and thin plates is also possible. A matter of interest is an application of liquid absorbers which concentration in the moderator or in a special water circuit may vary. In water-water power reactors a compensation of the reactivity by means of changing the neutron spectrum may appear to be difficult for its realization.

The following drives for solid operating devices of the control and safety system seem to be expedient.

I. Electromechanical or electromagnetic devices. They are to be placed in jackets welded in the reactor cover and to be connected with the operating device by a rigid rod through a remotely-disconnectable coupling. The jacket must be closed, that means the power supply of the drive is performed by an electromagnetic way through a closed non-magnetic partition.

While conducting a recharging the drive rods must be disconnected from the control assembly and removed upward but the control assembly should remain in an extreme low position. The absorber, protruded over the fuel core, and the rod of rigid connection may require a protection from the effect of water flow passing through the core.

Such a system of an electric-actuating drive assembled above the fuel core resembles those now existing and is accessible for inspections through the top flanges of the cover jackets. However, in principle, this system is unfit for in-run rechargings, because it does not ensure an emergency protection during the recharging without the cover removal, occupies a large area on the cover and requires a reserve space for an upward travel of the rods while removing them from the recharging space.

2. Hydraulic drive applied beneath the fuel core (Fig.7). A piston is connected with a "tail" of the control assembly and travels inside a hydrocylinder owing to the regulation of water pressure beneath the piston. The system contains assemblies of a disconnectable coupling of the piston with the "tail" of the assembly and the cylinder with the terminal of a water controller, assemblies of the extreme position limitation and fixation of intermediate positions, as well as an assembly of the lifting speed limitation.

Such a drive provides a prompt clearing of the space for recharging without its disconnection from the control assembly and with a maintenance of the emergency protection for the recharging period.

The hydraulic drive requires rather reliable connection of the piston with control assembly to prevent from a possibility of its upward motion under the force of the flow. This circumstance compels to seek for combined drives with an upper mechanically moved to rest and ensure fixation of intermediate positions. This structure is rather bulky but it provides a reliable and prompt downward travel of absorbers from any intermediate position.

The hydraulic drive can be combined with non-fuel operating devices of the control and safety system which are not endangered by an upward motion due to the effect of the cooling water flow, or an undetachable coupling of the hydraulic drive piston with the control assembly should be provided.

Problems of burn-up compensation by reduction of capacity

A matter of interest is a possibility of increasing the burn-up and reducing the cost of a kwh at the expense of a gradual decrease of the reactor capacity providing the maintenance of parameters with the aim of a burn-up compensation owing to a reduction of Doppler-effect (and, to a certain extent, a reduction of poisoning). This would reduce the fuel component of kwh cost but enhance the capital one because during a requiting period the power production will be lower than usual. That is why there is an optimum of the power reduction calculated below.

With a linear approximation the kwh cost can be represented as follows:

$$C_э(\varkappa) = C_T + C_K = C_T^\circ\left[1+k(1-\varkappa)\right]^{-1} + C_K^\circ\left[1-nk\ln\varkappa\right]\left[1+nk(1-\varkappa)\right]^{-1} \tag{3}$$

Where $C_T^\circ = C_T/\eta\rho_\omega^\circ$ and $C_K^\circ = \mathcal{K}/\tau\eta Q_0$ - are fuel and capital components of the kwh cost while operating at a rated capacity with a burn-up ρ_ω°; C_T is a cost of the I kg fuel elements; C_K° - is a total capital cost; $k = \Delta\rho_\omega^\circ/\rho_\omega^\circ \cong \Delta K_Q^\circ\,\rho_\omega^\circ(\partial K/\partial\rho_\omega)$ is a relative increase of the burn-up owing to application of the reactivity ΔK_Q° released during a reduction of the capacity down to zero; n is a number of rechargings during a core life τ_K with the capacity Q_0, $\varkappa = Q/Q_0 = \exp(-\tau_д/\tau_K k)$ is a value down to which the capacity will be reduced during additional time of the operation $\tau_д$

When n = I this cost $C_э(\varkappa)$ is less than $C_э(\varkappa=1) = C_э^\circ = C_K^\circ + C_T^\circ$ by a fraction $(\Delta C/C_э^\circ) = k\left[1-\varkappa+(1+C_T^\circ/C_K^\circ)^{-1}\ln\varkappa\right]\left[1+k(1-\varkappa)\right]^{-1}$ which in the case of $k(1-\varkappa) \ll 1$ has a maximum value when $\varkappa = \varkappa_{opt} = (1+C_T^\circ/C_K^\circ)^{-1}$. E.g., when $C_T^\circ = C_K^\circ$ the results would be as follows: $\varkappa_{opt} = .5$, $\tau_д = \tau_K k \ln\varkappa^{-1} = .69k\tau_K$ and $(\Delta C/C_э^\circ)_{max} = .15k/(1+0.15k)$. Then, when $k = .4$ a gain in the kwh cost will be ~ 5 per cent and a duration of the operation with a reduced capacity $\tau_д = .28\tau_K$

When the number of rechargings is greater a gain in the cost of a kwh will be lower. Within $n \to \infty$ instead of (3) the kwh - cost will be $C_э(\varkappa) = C_T^\circ\left[1+k(1-\varkappa)\right]^{-1} + C_K^\circ \ln\varkappa^{-1}/(1-\varkappa)$

which would be lower than $C_э^\circ = C_T^\circ + C_K^\circ$ only in the case of rather high k or C_T°/C_K°. E.g., when C_T°/C_K° = I the value $k = \Delta\rho_ш^\circ/\rho_ш^\circ$ is to be higher than .5.

Although the gain in the cost is not very high, it should be noted that this gain is to be obtained against the cost reflecting a continuous recharging as a neutron non-productive capture is excluded while compensating by absorbers.

Above all, the use of power effect simplifies the regulation system owing to a reduced number of devices compensating the burn-up and makes it possible to prolong the time between rechargings, i.e. to enhance a value of rechargings with a cover removal.

3. Optimization of water-water power reactor parameters

Upon selecting definite types of principal units for an atomic power station and a reactor, the optimization of parameters characterizing these units in numerical respect can give a proper orientation in the ways of the development and can essentially reduce the cost of a kwh.

As far as the number k of independent parameters x_k is high and their relations with the cost of energy are complicated, a direct optimization by means of a consistent solution of a closed system of equations (conditions for the minimum cost of a kwh $\partial c_э/\partial x_k = 0$) is possible only in extraordinary cases [6]. Often these conditions have a complicated form, because, while varying each parameter, all other parameters preselected remain unchanged and those "convenient" combinations which give a simple form of conditions $\partial c_э/\partial x_k = 0$ upon fixation do not remain constant. Providing such combinations selected, it would be possible to find out simple expressions for optimal parameters which would reduce the volume of calculations by many orders. In some cases it would be useful to fix, in particular, the reactor capacity as well as capacity and cost of a part of basic electromechanical equipment of the atomic power station.

Considering that relative changes in the cost of a kwh $C_э$ are usually considerably lesser than relative changes in independent parameters varied, an essential simplification can be also obtained.

Methods of these like were elaborated in the works [7] and utilized applicably to water-water power reactors of the type considered.

All the parameters have been subdivided into three groups. Parameters of the first and second groups in optimum are expressed analytically in terms of the third "initial" group of parameters. It includes coefficients of the equipment cost as well as a volumetric fraction of water in the core ε, the pressure p_1, coefficient irregularities of heat release temperature rese and water velocities, as well as other parameters which optimums have not been yet obtained in convenient analytical forms.

The first ("thermomechanical") group includes the capacity Q the length ℓ of the fuel core, a water temperature rise t in the core, a temperature difference in the steam generator and other parameters connected with listed above. These values can be optimized upon fixation of the fuel cost which is defined by the second group parameters.

The second ("fuel") group comprises the diameter d_T, a burnup $\rho_ш$ and the core life τ_K. They are interconnected by the definition $\rho_ш = \tau_K \rho_{sp}(d_T)$ through the specific power $\rho_{sp} = \frac{4q_f}{\gamma_u d_T} = \frac{4q_n}{\gamma_u \pi d_T^2}$.

A heat flow from the length unit q_n or surface unit q_f of the fuel element must be kept equal to ultimately admissible q_n° or q_f° in; when $q < q^\circ$ it is possible to gain a certain advantage by equalizing these flows, for example, by a corresponding increase of the fuel element diameter that would reduce the cost of the fuel element manufacturing and the number of their coatings the same capacity and otherways equal conditions provided.

When parameters of the first group (including the capacity, parameters of the coolant and that of the steam) are fixed, the capital investments and efficiency will be constant and the minimum cost of a kwh can be calculated with values ρ_w, τ_k and d_T minimizing the fuel cost.

Analysis of expression obtained by such a method in the works [7] for a cycle without repeated usage of the fuel results in the following:

I. In the field of diameters of fuel elements $d_T \geq d_o = q_n^{\partial}/\pi q_F^{\partial}$ x) where a linear heat flow is equal to the admissible $q_n = q_n^{\partial}$, the optimal burn-up ρ_{opt} is governed only by the dependence $C_{TH}(\rho_w)$ of the fuel cost on the burn-up and is maintained in the point $\rho_w = \rho_{opt}^{o}$ in which a tangent to the curve $C_{TH}(\rho_w)$ passes through origin (i.e. $\partial C_{TH}/\partial \rho_w = C_{TH}/\rho_w$)

In this case the optimal diameter is in proportion to the complex $(q_n^{\partial} K_{usr}/C_{TH}(\rho_w) \cdot \rho_w)^{1/4}$, i.e. it is to decrease with an increase of the burn-up, the fuel cost and with a reduction of the manufacturing cost a length unit of the fuel element ($.25\,\pi \gamma_v K_{usr}$) - where γ_v is a specific weight of the fuel.

This has created a trend to a transference of the optimum d_T to the field $d_T < d_o$.

2. In this field ($d_T \leq d_o$) the heat flow q_F equals the admissible q_F^{∂}, an optimal burn-up is somewhat lower than the value ρ_{opt}^{o} and it approaches to as the value $\sqrt{q_F^{\partial} / \frac{\partial C_{TH}}{\partial \rho_w}} \sqrt{K_{usr}}$ with an increase of q_F^{∂}, and with a decrease of the manufacturing cost of length unit of the fuel element.

And the optimal diameter in this case changes as $\sqrt{K_{usr}/(C_{TH} - \rho_w \partial C_{TH}/\partial \rho_w)}$.

3. Optimal value of ρ_w and d_T are susceptible to the dependence $C_{TH}(\rho_w)$ of the charged fuel cost on the extracted fuel burn-up. Apparently, the optimum for a dioxide fuel in some cases will be within the limits 30-50 kg per ton and an optimal diameter will not differ much from $d_o = q_n^{\partial}/\pi \cdot q_F^{\partial}$ = 10 mm.

The optimal core life is more stable. It has a value of about 10^3 days and grows as a complex $\sqrt{K_{usr}} / \sqrt{q_n^{\partial}(C_{TH}/\rho_w)}$, with decreasing of the permissible heat flows and a ratio (C_{TH}/ρ_w), together with an increase of the manufacturing cost.

With the help of corresponding equation given in the works [7] the calculations have been made for 288 combinations of optimal first group parameters proceeding from different initial parameters of the third group which contain some conditionalities and due to which they have varied within wide limits.

The following has been taken into consideration while calculating:

I. Maximum enthalpy i_H of water in the outlet of the assemblies should not essentially differ from the saturation enthalpy $i_H = i'(P = P_1)$. When $i_H > i'$ it should be necessary to reduce the enthalpy in the inlet i_{BX} or to increase the water flow rate (in both cases the efficiency will drop) or to reduce the capacity. These negative effects cannot be compensated by a certain increase of critical heat fluxes with the increase of water subcooling in the dangerous region located higher than the mean part of the fuel core.

When $i_H > i'$ a steam quality $x = (i_H - i')/r$ is to occur that worsens the neutron balance and reduces the hydraulic stability of a parallel operation of the channels; a steam condensation upon mixing causes hydraulic shocks. With a low x these effects are apparently inessential but the numerical aspect of the problem has not been yet clear.

x) Here $d_o = q_n^{\partial}/\pi q_F^{\partial}$ is a diameter with which the heat flows from a length unit and from a surface unit of the fuel element are simultaneously equal to those permissible: $q_n = q_n^{\partial}$ and $q_F = q_F^{\partial}$

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2. The diameter of the active core should be selected close to the maximum value (3m) of a core fit for a factory-manufactured vessels of ultimate transportable size ($D_{tp} \cong 4$m).

The use of a diameter larger than D_{tp} should be advantageous if it would not be accompanied by a reduction of the thickness and admissible strength of the vessel material and by a corresponding sharp drop of the pressure and efficiency which cannot be compensated by an increase of the capacity (in boiling reactors the matter might be somewhat different).

A decrease of the diameter causes a fast drop of the capacity which cannot be compensated by the reduction of the vessel cost and by an increase of the efficiency with the increase of the pressure permissible which, with D=4m is close to optimal one. A gradual increase of the pressure provided, the efficiency increase develops slowly and critical heat fluxes drop more quickly that compels either to reduce the capacity or to increase the surface of the fuel elements. When $\partial \eta_{epui} / \partial p_1 = -.03 \cdot 10^6 \frac{\text{kcal}}{\text{cu m hr} \cdot \text{abs atm}}$ (see [8]), the optimal pressure will be approximately 120 abs atm that is yet admissible for a transportable vessel. A problem of another kind is an achievement of sub- and supercritical parameters which can sharply change many properties to the better. But this problem is not considered here.

The results of applicating the equations calculated are chiefly represented by the following provisions which are valid in the point of the minimum kwh cost. (They, as a rule, allow rather clear physical interpretation, though sometimes the "common sense" could prompt some erroneous trends.)

1. An effect of the water-uranium ratio N_v on the cost capital component of the kwh cost in optimum is rather low in spite of a big change of the fuel element surface and the coolant flow area while changing N_v; this proceeds from the compensation of an effect of N_v by a reverse effect of the relative length of the fuel core (l/d_1 which promptly increases in optimum upon increase of N_v (approximately as $N_v^{0.6}(1+N_v)^{0.4}$

2. An optimal thermal capacity Q_{opt} mainly depends on the radial coefficient of a heat release irregularity K_q and equals 5 ÷ 6 millions of thermal kw when K_q =1.5 and 3.3 ÷ 4 millions kw when K_q = 3. Approximately $Q_{opt} \sim K_q^{-0.6}$ and practically it does not depend on permissible heat flows and it seems to be advantageous to utilize an increase of these flows for a reduction of the fuel core relative length.

3. An average temperature rise of the coolant in optimum chiefly depends on the heating irregularity $K_{\Delta t} = \frac{\Delta t_{max}}{\Delta t_{aver.}}$ (approximately, $\Delta t_{opt} \sim K_{\Delta t}^{-1}$) and makes up 45÷65°C when $K_{\Delta t}$ = 1.2 and 30÷45°C when $K_{\Delta t}$ =2.

4. An optimal average water velocity in the core W_{opt} is mainly governed by the speed irregularity $K_w = W_{max}/W_{aver.}$ and makes up 3.5÷4 m p sec when K_w =2.5 and 6÷7m p sec when K_w = 1.25 (approximately, $W_{opt} \sim K_w^{-0.8}$).

Maximum temperature rise Δt_{max} and velocity W_{max} especially are more stable than mean ones; E.g., $W_{max} = K_w W_{opt}$ makes up 8-9 m p. sec. approximately.

While changing the power density ρ removal from a coolant volume unit, it is useful to change the velocity and heating of water in one and the same rate as $\sqrt{\rho}$

5. An optimal fraction of an energy consumption for pumping the coolant $n_v = N_v/Q$ rather weakly depends on all the initial data varied and makes up (.6-.9) per cent of the heat capacity.

6. An optimal mean temperature difference in the steam generator is also stable and for a stainless heating surface it makes up approximately 25°C.

7. And, at last, the unit cost C_{yer} (investments per 1 kw without the first charging cost) is chiefly governed by the dependence of the second circuit cost and that of auxiliary systems of the atomic power station on the capacity: $C_{AЭС} = A+BQ$. The

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main part is played by the coefficient B, but it is interesting that it has rather low effect on the thermomechanical parameters optimized.

A unit cost is essentially reduced upon the reduction of irregularity coefficients K_q and $K_{\Delta t}$

Upon an increase of the pressure the value of the unit cost drops too but the length of the fuel core as well as the fuel charging are increased (due to a decrease of permissible heat flows). That is why there is an optimum of the pressure p_1 of about I20 abs atm which was spoken about previously. With this pressure and a maximum water enthalpy corresponding to the saturation for a mean heat flow on the surface of the most intense fuel element $q_M = I.2 \cdot I0^6 \frac{kcal}{m^2 \cdot ^oC}$ when the radial irregularity of heat release $K_q = I.5$, the optimal parameters of a water-water power reactor with a maximum-size transportable vessel of 4m in diameter are laid approximately within the following limits: the thermal capacity $Q = 5.5 \cdot I0^6$ kw, the electrical capacity $N = I.6 \cdot I0^6$ kw, the heat removal per the collant flow area unit $\sim I.3 \cdot I0^6 \frac{kw}{sq\ m}$, the fuel core length-fuel element diameter ratio $\sim$ 500 (when water-uranium ratio $\sim$ I.6), the mean heating and velocity of water 48°C and 6.5m p sec (when $K_{\Delta t}$ = I.2 and K_w = I.25); the mean temperature difference in the steam generator 26°C; the steam pressure p_2 = 45abs atm, the cycle efficiency $\sim$30 per cent, a fraction of the power consumption for the circulation $\eta_4 \simeq .65$ per cent (when the pump efficiency $\sim$ .7).

Parameters of the second unit of the Novo-Voronezh atomic power station are close to optimal values as to the pressure, coefficients of the irregularity and reactor capacity (Q = I400 Mw) which is in the second unit yet rather lower than an optimal one.

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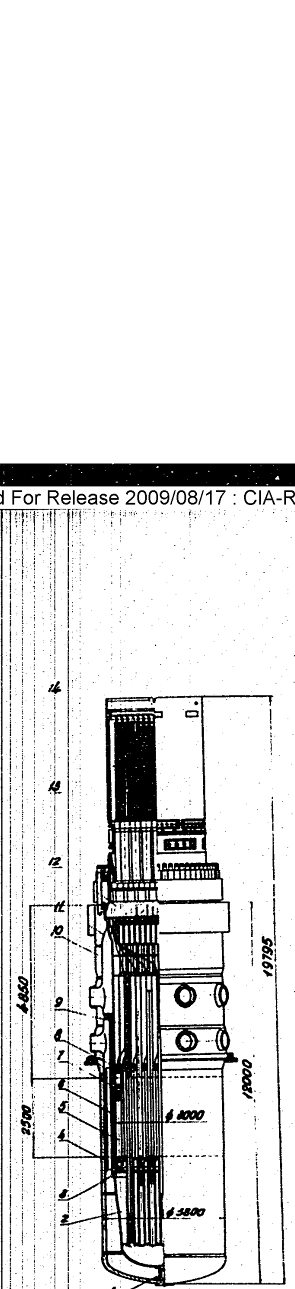


Fig.1 General view of the reactor

1 - Shaft to vessel clutch sleeve; 2.- Control assembly; 3 - Bottom of the shaft; 4 - Shield; 5 - Fuel assembly; 6 - Shaft; 7 - Core tank (removable); 8 - Spacing grid; 9 - Protective tube assembly; 10 - Pressure vessel; 11 - Communications of the temperature control system; 12.- Removable cover block; 13 - Control assembly drive; 14 - Maintenance area

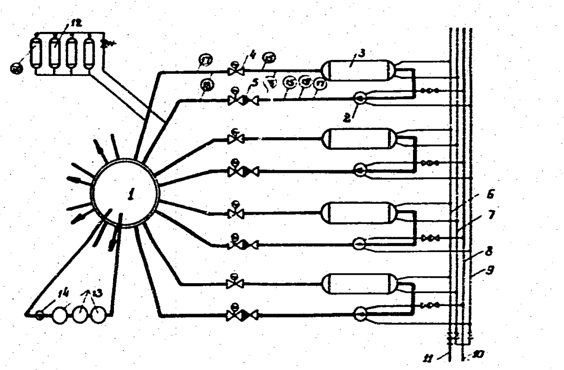
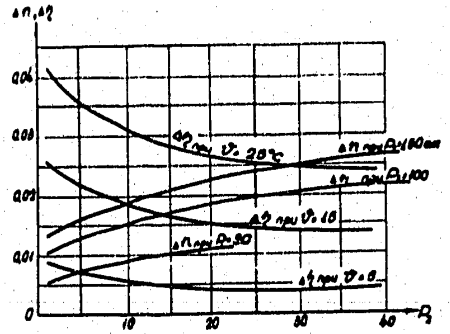


Fig.2 Reactor cooling scheme

1 - Reactor; 2 - Main pump; 3- Steam generator 4- Valve; 5 - Non-return valve; 6. Pipeline of air removal; 7 - Drain pipeline; 8 - Pipeline of air removal out of pumps; 9 - Pump drain; 10 - Pipeline to drainage tanks; 11 - Pipeline to air cleaning filters; 12 - Pressurizers; 13 -Filters 14 - Auxiliary circulation pump; 15 - Temperature sensing element; 16 - Sensing element of temperature rise in the loop; 17 - Pressure transmitter; 18 - Element of the pressure drop on the core; 19 - Flow rate element; 20 - Level meter in the compensator



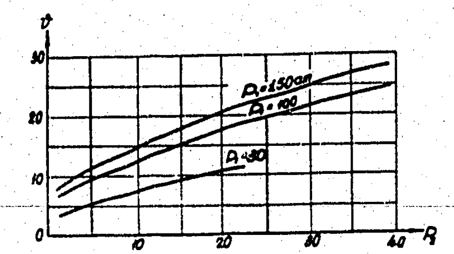


Fig. 3 The increase of the thermodynamic efficiency $\Delta\eta$ and of the relative pump power Δn (when steam generator is replaced by a throttling installation) as a function of the temperature head in steam generator ϑ and of the pressure in the reactor p_1 and on the turbine p_2.
The lower plot shows the relation between minimum temperature head ϑ and p_1 and p_2 pressures, at which that is net efficiency does not change.

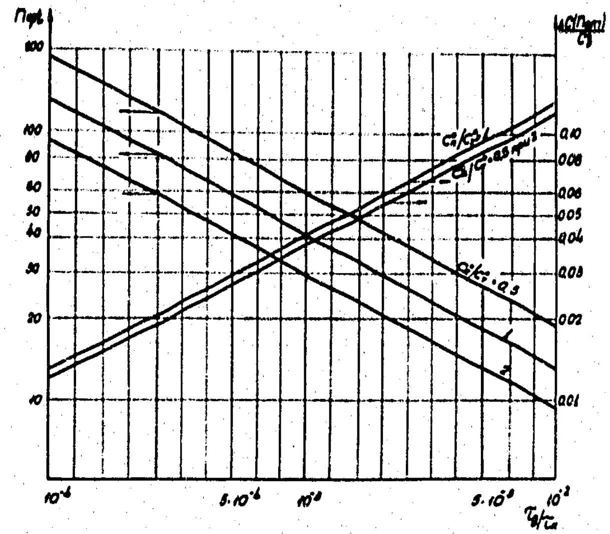


Fig.4. The optimal reloading number n_{opt} for the core life τ_K and the corresponding relative excess of 1 kWh cost $\Delta C(n_{opt})$ (over the 1 kWh cost c_3^0 at continuous overload at full power), as a functions of the auscillary reloading time τ_B.

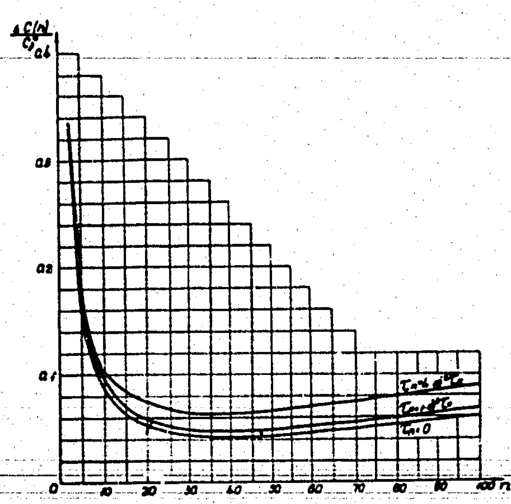


Fig.5. Relative excess of kWh cost $\Delta C(n)/C_3^0$, over kWh cost at continuous overload at full power, as a function of the reloading number n for core life τ_K and of reloading time τ_n (auxillary time taken equal to $\tau_B = 10^{-3}\tau_K$).

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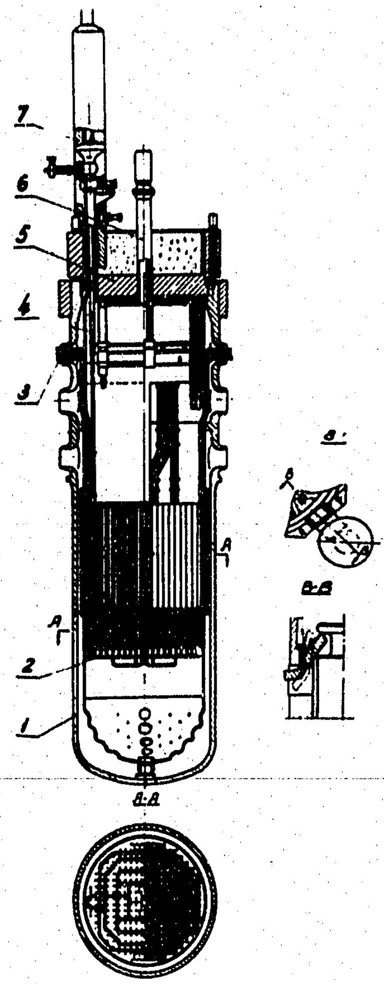

Fig.6 Reactor scheme with "honeycomb" core and with the change of liquid absorber concentration in the channel system

1.- Pressure vessel; 2 - Collectors of liquid absorber; 3 - Pipes of fuel assemblies hermeticity control and water temperature control; 4 - Cask mechanism rod; 5 - Cover; 6 - The reloaging machine; 7 - Cask; 8 - Unit of fuel assembly fixation from floating

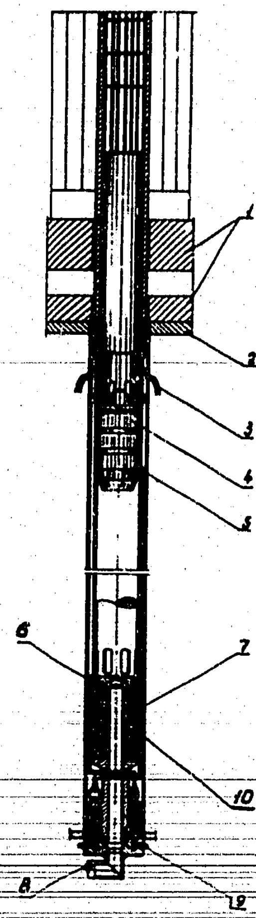

Fig.7 The scheme of hydrodrive for control assembly

1 - Support plate; 2 - Plate with hydrocylinder jackets; 3 - Inlet of the cooling water; 4 - Piston-shank of the operating organ(upper position); 5 and 6 - Position indicators; 7 - Damper pipe; 8 - Water supply under the piston; 9 - To position indicators 10 - Piston in the down position

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